Satisfaction of Appendix K Requirements for the NuScale Power Module

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Revision 0

Nonproprietary

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CONTENTS

TABLES

FIGURES

1.0 Introduction

Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, in 10 CFR Part 50 (Reference 1.4.1), requires that light water nuclear reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding be provided with emergency core cooling systems (ECCS) that are designed in such a way that their calculated core cooling performance after a postulated loss-of-coolant accident (LOCA) conforms to certain criteria specified in paragraph 50.46(b). Paragraph 50.46(b)(1) requires that the calculated maximum temperature of fuel element cladding not be greater than 1204.4 °C (2200 degrees °F). In addition, Paragraphs 50.46 (b)(2) through (b)(5) contain required limits for calculated maximum cladding oxidation and maximum hydrogen generation, as well as the requirements that calculated changes in core geometry remain amenable to cooling and that long-term decay heat removal be provided. This requires that ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of cases to provide assurance that the most severe postulated loss-of-coolant accidents are identified.

Two options, paragraph 50.46(a)(i) and paragraph 50.46(a)(ii), for ECCS performance calculation are allowed for providing the assurance that the most severe design-basis accident has been evaluated. Paragraph 50.46(a)(i) requires that the best-estimate approach detailed in Regulatory Guide 1.157 (Reference 1.4.2) be followed, and paragraph 50.46.(a)(ii) endorses the conservative deterministic approach detailed in Appendix K.

Best Estimate Approach

The evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. Best-estimate evaluation models are not considered further in this white paper.

Appendix K Approach

An ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K (Reference 1.4.3) ECCS evaluation models.

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The NuScale Power Module is a natural circulation integral pressurized-water reactor (PWR) with passive safety systems. The plant is designed to greatly reduce the consequences of design-basis LOCAs. Consequently, many of the phenomena that are the subject of Appendix K requirements are not encountered in the design-basis LOCA for the NuScale design. That is, only a subset of the phenomena that are addressed in Appendix K will be encountered in design-basis LOCAs due to the simple and safe design of the NuScale Power Module.

For example, from the simplicity perspective there are no main coolant pumps in the NuScale Power Module. Therefore, the phenomena that are the subject of Appendix K, Requirement 6, Pump Modeling, will not be encountered. Likewise, from the safety margin perspective limitations on clad temperature and differential pressure are established that provide sufficient margin to

preclude the need for a clad rupture model required by Appendix K, Item B, Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters. The small amount of cladding swelling that might occur will not result in rupture and will be modeled with the variable gap conductance model.

1.1 Purpose

The purpose of this report is to describe the approach that NuScale shall take to meet the requirements of Appendix K to 10 CFR 50 (Reference 1.4.3). The manner of addressing each Appendix K requirement is addressed in Section 3.2, with emphasis placed on those requirements where limitations will be imposed.

1.2 Scope

This report describes the NuScale plant, with emphasis on ECCS, and addresses each requirement of 10 CFR 50, Appendix K, and it describes the method of complying with that requirement. The manner of addressing each Appendix K requirement is addressed in Section 3.2, with emphasis placed on those requirements where compliance is achieved by imposing limitations on the use of the code. The code modifications and additions for NuScale plant-specific features are also addressed, and preliminary scoping analyses with Appendix K are presented.

1.3 Abbreviations and Definitions

Table 1-1. Abbreviations

Term	Definition
BWR	boiling water reactor
CCFL	counter-current flow limitation
CHF	critical heat flux
CVCS	chemical and volume control system
DHRS	decay heat removal system
ECCS	emergency core cooling system
EM	evaluation model
EMDAP	evaluation model development and process
FLECHT	full-length emergency cooling heat transfer
HCSG	helical coil steam generator
HTFFS	heat transfer and fluid flow service
INL	Idaho National Laboratory
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
MIT	Massachusetts Institute of Technology
NIST	NuScale Integral System Test
NRC	Nuclear Regulatory Commission

Term	Definition
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PWR	pressurized-water reactor
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RVV	reactor vent valve
SBLOCA	small-break loss-of-coolant accident
SIET	Societa Italiana Esperienze Termoidrauliche
SMR	small modular reactor
THTF	thermal-hydraulic test facility

1.4 Referenced Documents

1.4.1 *U.S. Code of Federal Regulations*, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” Section 50.46, Part 50, Title 10, “Energy,” (10 CFR 50.46).

1.4.2 U.S. Nuclear Regulatory Commission, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” Regulatory Guide 1.157, May 1989.

1.4.3 *U.S. Code of Federal Regulations*, “ECCS Evaluation Models,” Appendix K, Part 50, Title 10, “Energy,” (10 CFR 50, Appendix K).

1.4.4 U.S. Nuclear Regulatory Commission, “Transient and Accident Analysis Methods,” Regulatory Guide 1.203, December 2005.

1.4.5 The RELAP5-3D Code Development Team, “RELAP5-3D Code Manual Volume I: Code Structure, System Models and Solution Methods,” INEEL-EXT-98-00834, Revision 4.0, June 2012.

1.4.6 NuScale Power, LLC, “Gap Analysis Summary Report,” NP-RP-0612-023, Revision 0, July 2012.

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1.4.8 U.S. Nuclear Regulatory Commission, “Testing and Computer Code Evaluation,” AP1000 Final Safety Evaluation Report, Chapter 21, Section 21.6.2, p. 87, Agencywide Document Access and Management System (ADAMS) Accession No. ML033290640.

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1.4.18 McDonough, J.B., W. Milich, and E.C. King, “An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube,” Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197-208, (1961).

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2.0 Plant Description

The NuScale SMR plant consists of one or more NuScale Power Modules, each of which is a small 160 MWt, natural circulation reactor with passive PWR. A single NuScale Power Module consists of four major components:

- the nuclear steam supply system (NSSS), which includes the nuclear core; the helical coil steam generators, and the pressurizer, which are all within a single pressure vessel as well as the safety relief valves and reactor venting and recirculation valves that connect to the pressure vessel
- a standard steam power conversion system, which includes the steam turbine-generator, the main condenser, and the feedwater system
- the steel containment vessel that houses the NSSS and resides in the reactor pool
- the submerged heat exchangers, piping, and valves of the decay heat removal system (DHRS)

The NSSS module is designed to operate efficiently under full power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps. As shown in Figure 2-1, the reactor core is located inside a shroud connected to the hot leg riser. The reactor core heats reactor coolant causing the coolant to flow upward through the riser. When the heated reactor coolant exits the riser, it passes the tubes of the helical coil steam generator, which act as a heat sink. As the reactor coolant passes through the steam generator, it cools, increases in density, and naturally circulates down to the reactor core, where the cycle begins again.

Nuclear steam supply system modules are completely submerged in a reactor pool and protected by passive safety systems. Each NSSS module has a dedicated chemical and volume control system (CVCS), ECCS, and DHRS.

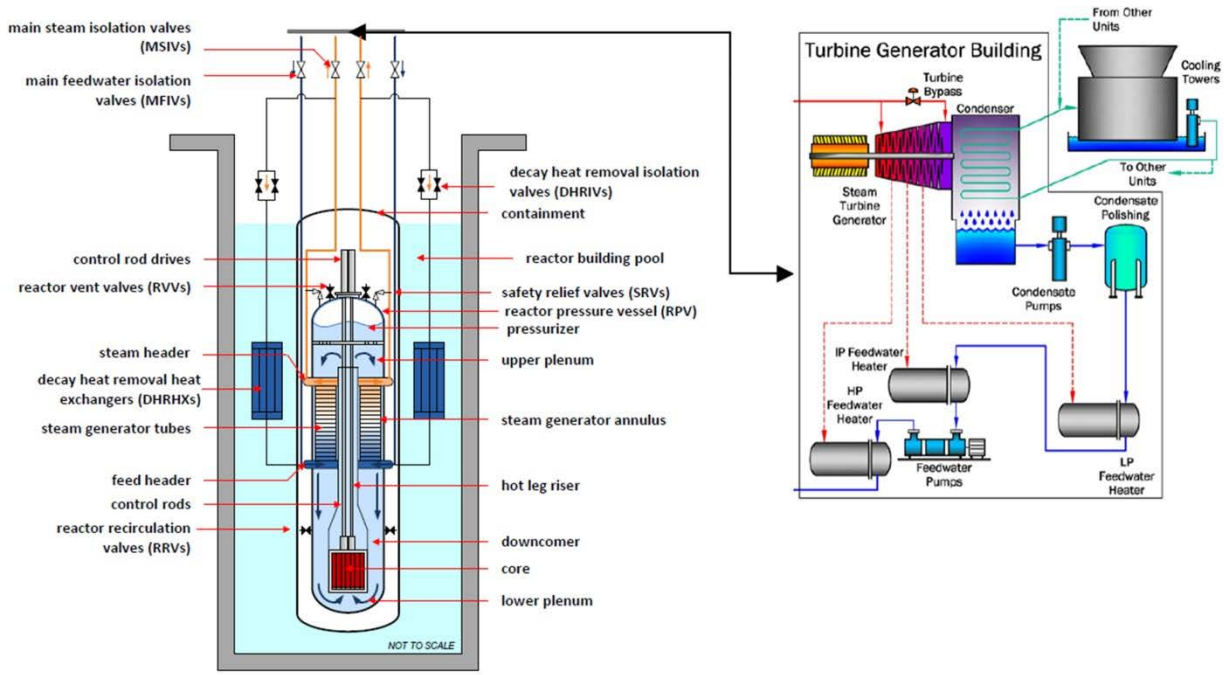

Figure 2-1. Schematic of a single NuScale Power Module

2.1 Description of NuScale Emergency Core Cooling System and Decay Heat Removal System

The NuScale reactor was designed to eliminate large-break loss-of-coolant accidents (LBLOCAs) and minimize the effect of postulated small-break loss-of-coolant accidents (SBLOCAs). All small-break LOCAs are mitigated by actuation of the ECCS. As shown in Figure 2-2, the ECCS consists of two independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). The ECCS is initiated by opening the two RVVs in lines that exit the top of the reactor pressure vessel (RPV) and the two RRVs on lines entering the RPV in the downcomer region at a height above the core. Opening the valves allows a natural circulation path between the RPV and containment to be established. Water that is vaporized in the core leaves through the RVVs as two-phase steam and water during the blowdown phase, and single phase steam during the long-term cooling phase. It is then condensed and collected in the containment vessel, and is returned to the downcomer region inside the RPV through the RRVs. This natural circulation loop removes decay heat from the core in the RPV and deposits it in the containment. Heat deposited in the containment is then transferred by conduction, convection, and boiling to the water in the reactor pool.

The DHRS provides an additional capacity to remove decay heat during the initial blowdown period of a LOCA and at least the first three days of long-term cooling following the accident. However, the DHRS is not required, nor credited, for such events. The DHRS provides secondary side reactor cooling when normal feedwater is not available. The system, as shown in Figure 2-2, is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each steam generator loop. Each train is independently capable of removing 100 percent of the decay heat load. Each train has a passive condenser submerged in the reactor pool. The condensers are maintained with sufficient water inventory for stable operation.

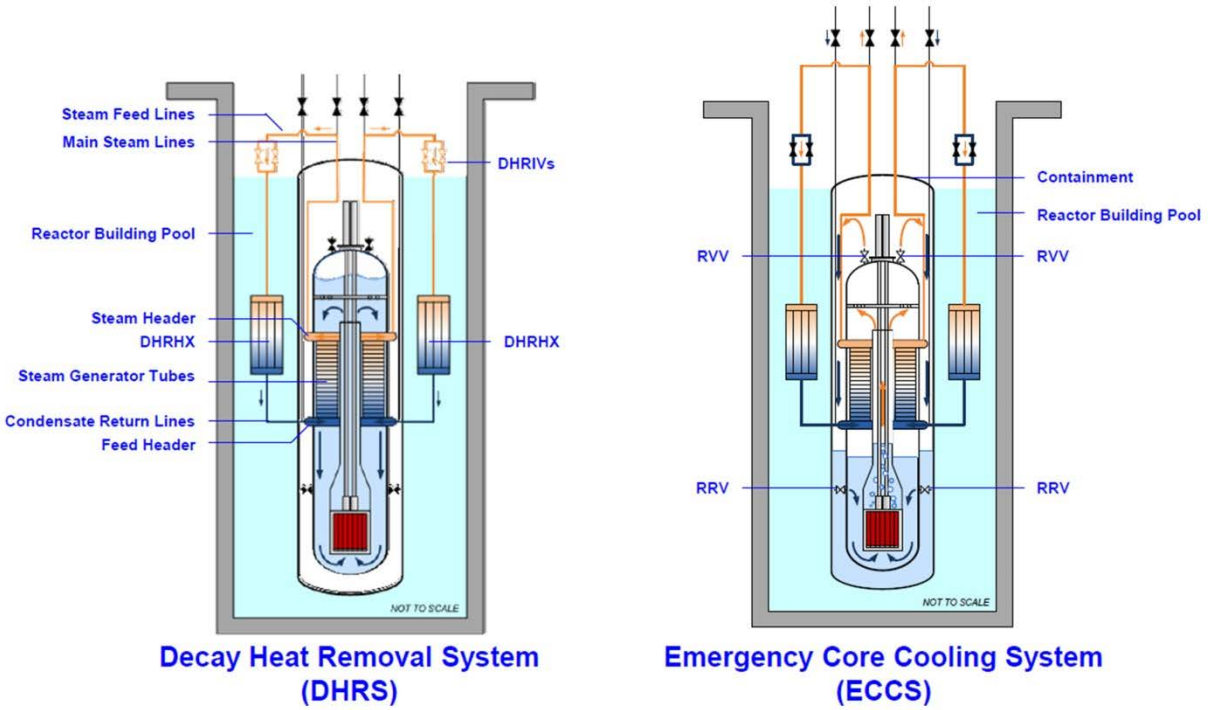

Figure 2-2. Schematic of NuScale Power Module DHRS and ECCS

2.2 Emergency Core Cooling System Evaluation Model Development and Assessment Process

The Evaluation Model Development and Assessment Process (EMDAP), described in Reference 1.4.4, establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. The EMDAP described here has been developed for simulating small-break LOCAs in the NuScale SMR Power Module. While there are differences between the NuScale SMR and current generation PWRs, many of the basic physical phenomena are the same. Also, the NuScale LOCA evaluation model (EM) utilizes the NRELAP5 code that was developed based on the Idaho National Laboratory (INL) RELAP5-3D computer code (Reference 1.4.5).

By design, the margins to LOCA regulatory limits for the NuScale SMR are much greater than for current generation PWRs, so that a number of phenomena that occur and are important in a current generation PWR SBLOCA do not occur in the NuScale SMR. Examples of phenomena that do not occur for NuScale SMR design-basis SBLOCAs include loop seal clearing, pump coastdown, two-phase pump performance, entry of significant amounts of noncondensable gases into the system, core reflooding, clad swelling and rupture, metal-water reaction and ECCS bypass. (A more detailed examination of differences and a detailed proposed reconciliation of existing light water reactor [LWR] regulatory requirements and guidance with the characteristics of the NuScale reactor plant design is presented in the gap analysis summary report [Reference 1.4.6]).

Differences due to geometry and operating mode between current PWRs and the NuScale SMR include the importance of natural circulation in multiple loops when DHRS is operational, a high pressure containment, heat transfer to a reactor cooling pool, and helical coil steam generator performance. {{

}}3(a)-(b)

Loss-of-coolant accidents are postulated breaks in the reactor coolant pressure boundary that result in leakage of reactor coolant at a rate exceeding the capability of the normal reactor coolant makeup system. Breaks of various sizes, types, and orientations are postulated to occur in piping carrying primary system coolant from and to the reactor vessel. Since there are no large pipes connected to the NuScale reactor pressure vessel, all breaks are at most small LOCA events. With the elimination of most primary coolant carrying piping, small breaks in the NuScale design are limited to reactor safety valves, ECCS valves, charging and letdown lines, pressurizer spray lines and instrumentation lines. Loss of significant quantities of coolant inventory from the reactor vessel could adversely affect heat removal from the reactor core. The 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants (Reference 1.4.7) requires each PWR to be equipped with an ECCS that provides emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that fuel and clad damage that could interfere with continued effective core cooling is prevented and clad metal-water reaction is limited to negligible amounts. The NuScale reactor does not need an ECCS safety injection system due to the absence of large break LOCAs and a direct access to the ultimate heat sink (i.e., the reactor pool). Water from the primary system is captured in containment and returned to the reactor core using ECCS valves during SBLOCAs. {{

}}3(a) The heat generated in the RPV is transferred to the reactor pool through the containment vessel. Natural circulation between the core and the containment is a cooling mechanism from the core to the containment. This cooling mechanism is maintained for long-term cooling for all conditions and events without the need for supplying makeup water to the RPV or containment.

The purpose of the EM analysis is to demonstrate that the ECCS system is capable of mitigating the consequences of the full spectrum of design-basis LOCAs in the NuScale SMR power module. The EM will be developed to demonstrate compliance with ECCS acceptance criteria 10, CFR 50.46 using a deterministic approach that implements the modeling requirements of 10 CFR 50, Appendix K. Models are being added to the NRELAP5code, experimental tests will be conducted, and assessments will be performed to specifically address the unique requirements of the NuScale Power Module for LOCA analysis. As described in Section 3.6.4 below this includes integral and separate effects testing and code assessment at the NuScale Integral Systems Test (NIST) facility at Oregon State University (OSU), experiments at SIET in Italy on a near full-scale helical coil steam generator, high pressure containment condensation tests atNIST and full-scale rod bundle Critical Heat Flux experiments at Stern Laboratories in Canada.

3.0 Appendix K Compliance Evaluation

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}}3(a)-(b)

3.1 Emergency Core Cooling System Evaluation Model

The NuScale SBLOCA evaluation methodology is being developed following the guidelines in the evaluation model development and assessment process (EMDAP) of Regulatory Guide 1.203 (Reference 1.4.4). The EM specifically addresses the evaluation of the NuScale Power Module to meet the requirements of 10 CFR 50, Appendix K for evaluation against the acceptance criteria of 10 CFR 50.46 (Reference 1.4.1).

The NuScale LOCA EM will utilize the NRELAP5 code that was developed based on the INL RELAP5-3D computer code (Reference 1.4.5). A number of code features are being added or modified to enable Appendix K compliance. For example, the Moody critical flow model required by Appendix K will be added to the code, and heat transfer correlations applicable for helical coil steam generators will be added to address Appendix K, Section I.A.7, Pressurized Water Reactor Primary-to-Secondary Heat Transfer, that requires heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account.

3.2 10 CFR 50 Appendix K Compliance

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}}3(a)-(b)

Table 3-1. Appendix K required and acceptable features compliance

Appendix K Section	Required and Acceptable Feature	{{
A. Sources of Heat During the LOCA	For the heat sources listed in A.1-A.4 of this table, it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core life time must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.	
1. The Initial Stored Energy in the Fuel	The steady state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.)	
2. Fission Heat	Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.	
3. Decay of Actinides	The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties.	
4. Fission Product Decay	The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the October 1971 ANS standard (Reference 1.4.11)	}}3(a)-(b)

Appendix K Section	Required and Acceptable Feature
5. Metal—Water Reaction Rate	The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less than 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.
6. Reactor Internals Heat Transfer	Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.
7. Pressurized Water Reactor Primary-to-Secondary Heat Transfer	Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account.
B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters	Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.
C. Blowdown Phenomena	
1. Break Characteristics and Flow	
a.	In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall

{{ }}[3(a)-(b)]

Appendix K Section	Required and Acceptable Feature	{{
	also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.	
b. Discharge Model	For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model.	
c. End of Blowdown (Applies Only to Pressurized Water Reactors)	For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall be subtracted from the reactor vessel calculated inventory in the calculations.	
d. Noding Near the Break and the ECCS Injection Points	The noding in the vicinity of, and including, the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.	
2. Frictional Pressure Drops	The frictional losses in pipes and other components, including the reactor core, shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident.	
3. Momentum Equation	The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.	
4. Critical Heat Flux		
a.	Correlations developed from appropriate steady state and transient-state	}}3(a)-(b)

Appendix K Section	Required and Acceptable Feature	{{
	experimental data are acceptable for use in predicting the CHF during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.	
b.	Steady state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following: (six acceptable CHF correlations are listed)	
c.	Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF that allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used.	
d.	Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following: (GE transient CHF correlation is listed).	
e.	After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location, subsequently during the blowdown, even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.	
5. Post-CHF Heat Transfer Correlations		
a.	Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.	

}}3(a)-(b)

Appendix K Section	Required and Acceptable Feature	{{
b.	The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime," AECL-3281, revised December 1969) (Reference 1.4.16) and the Westinghouse correlation of steady state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," USNRC Docket RM-50-1, page 25-1, October 26, 1972) (Reference 1.4.17) are acceptable for use in the post-CHF boiling regimes. In addition, the transition boiling correlation of McDonough, Milich, and King (J.B. McDonough, W. Milich, E.C. King, "An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube," Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197-208, (1961) (Reference 1.4.18) is suitable for use between nucleate and film boiling. Use of all these correlations is restricted as follows: 1. The Groeneveld correlation shall not be used in the region near its low pressure singularity 2. The first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300°F 3. Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300°F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.	
c.	Evaluation models approved after October 17, 1988, which make use of the Dougall-Rohsenow flow film boiling correlation, may not use this correlation under conditions where nonconservative predictions of heat transfer result.	
6. Pump Modeling	The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer	}}3(a)-(b)

Appendix K Section	Required and Acceptable Feature	{{
	between the fluid and the rotating member, with variable pump speed as a function of time. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data.	
7. Core Flow Distribution During Blowdown		
a.	The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of the cladding swelling or rupturing. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).	
b.	A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.	
D. Post-Blowdown Phenomena; Heat Removal by the ECCS		
1. Single Failure Criterion	An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation, the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.	
2. Containment Pressure	The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.	

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Appendix K Section	Required and Acceptable Feature	{{
3. Calculation of Reflood Rate for Pressurized Water Reactors	The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers, if this assumption leads to the maximum calculated cladding temperature; otherwise, the pump rotor shall be assumed to be running free. The ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (full-length emergency cooling heat transfer) (FLECHT). The effects on reflooding rate of the compressed gas in the accumulator, which is discharged following accumulator water discharge, shall also be taken into account.	
4. Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors	The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes, unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption.	
5. Refill and Reflood Heat Transfer for Pressurized Water Reactors		
a.	For reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results ("PWR FLECHT Final Report," Westinghouse Report WCAP-7665, April 1971) (Reference 1.4.19). The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied; presently available FLECHT heat transfer correlations "PWR FLECHT Group I Test Report," Westinghouse Report WCAP-7544, September 1970 (Reference 1.4.20); "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972) (Reference 1.4.21) are not acceptable. Westinghouse Report WCAP-	}}3(a)-(b)

Appendix K Section	Required and Acceptable Feature	{{
	7665 has been approved for incorporation by reference, by the Director of the Federal Register. A copy of this report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852-2738. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.	
b.	During refill and during reflood, when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture, as such blockage might affect both local steam flow and heat transfer.	
6. Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling		
7. The Boiling Water Reactor Channel Box Under Spray Cooling		}}3(a)-(b)

3.3 Code Features Modified to Enable Appendix K Compliance

RELAP5-3D has been widely used for LOCA analysis so most of the necessary models and correlations are in the code and have been subject to assessment by the worldwide user community. However, the RELAP5 family of codes was initially developed as best estimate LOCA codes, so some features required by Appendix K are not in the RELAP5-3D code. Also, the NuScale plant has design features that require additional or modified capabilities.

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3.4 SBLOCA EM Limitations

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3.5 Appendix K Requirements that are not Applicable to the NuScale Design

Following is a list of phenomena that do not occur in the NuScale power module, and hence, need not be addressed by the NuScale ECCS EM.

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3.6 Appendix K Evaluation Model Documentation Requirements

Section II, Required Documentation, of Appendix K requires that the applicant provide complete documentation of the EM in the following specific areas:

3.6.1 Description of the Evaluation Model

The existing RELAP5-3D code documentation consisting of five volumes will be provided along with an addendum that describes the model changes implemented by NuScale to produce the NRELAP5 computer code. The addendum will have the same level of detail as the RELAP5-3D code manuals. A complete listing of the frozen NRELAP5 computer program will be furnished to the NRC.

3.6.2 Convergence Sensitivities

Solution convergence will be demonstrated for the sample case to be included in the LOCA Methodology Topical Report. Convergence will be established with respect to system modeling, including nodalization and calculational time step size.

3.6.3 Modeling Sensitivities

Sensitivity studies will be performed to evaluate the effect on the calculated results of variations in modeling options, phenomena included in the modeling, and values of important model parameters. For items for which results are shown to be sensitive, the choices made will be justified.

3.6.4 Code Assessment against Experimental Data

NuScale is conducting a number of design-specific experiments to obtain data to validate the NRELAP5 model for LOCA and transients analysis. Integral systems and separate effects experiments are being performed in the NuScale Integral System Test (NIST) facility, {{ }}[3(a)] In addition to a number of integral LOCA tests, separate effects tests are planned for high pressure containment condensation and DHRS system performance.

Experiments are also being conducted at the Societa Italiana Esperienze Termoidrauliche (SIET) Laboratories in Piacenza, Italy. Tests will be conducted in a three-tube electrically heated facility and in a near full-scale helical coil steam generator (HCSG) thermal-hydraulic test facility. Testing will be conducted in full-length tubes for a range of fluid and flow conditions to obtain primary and secondary side thermal-hydraulic data for a full-length NuScale HCSG under prototypic operating conditions. The data will support plant engineering design in addition to steady state and transient safety analyses.

Critical Heat Flux tests were performed under steady state conditions by Stern Laboratories of Hamilton, Ontario, Canada. Both uniform and a cosine axial power shape were used for the axial power distribution and a center-peaked shape was used for the radial power distribution. To consider the cold wall effect, cases with and without a thimble tube were tested. The tests were conducted over a range of pressures, mass flux, heat flux, and inlet subcooling that covers the entire applicable range for the NuScaleplant. The CHF correlation developed using the Stern data will be used in the NRELAP5 EM, and in the NuScale specific sub-channel code.

In addition to the design-specific tests described above, the extensive separate effects data from experiments, including the semi-scale facility, ORNL THTF, Neptunus, CIRCUS, GE level swell, and other facilities are being used to establish the applicability of the NRELAP5 code to the NuScale design. A summary of the developmental assessment matrix for the NRELAP5 code is included as Appendix A.

A regression suite is also run to evaluate the effect of changes in NRELAP5 on the results for the extensive RELAP5-3D developmental assessment matrix.

4.0 Code Modifications for NuScale-Specific Plant Features

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5.0 Preliminary Scoping Calculation Results

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6.0 Results and Conclusions

NuScale shall perform the required ECCS analysis using an ECCS evaluation model developed in conformance with the requirements of Appendix K to 10 CFR 50, with the acceptance criteria of 10 CFR 50.46 (Reference 1.4.1). This is the second option discussed in 10 CFR 50.46(a)(1) and is acceptable per Regulatory Guide 1.157 (Reference 1.4.2). This white paper describes details of the approach that NuScale intends to take to meet the requirements of Appendix K to 10 CFR 50.46.

The NuScale Power Module is a natural circulation integral PWR with passive safety systems. The plant and the safety systems are designed to greatly reduce the consequences of design-basis LOCAs. While no exemptions to any of the Appendix K requirements are needed, many of the phenomena that are the subject of Appendix K requirements, while applicable, are not encountered in the design-basis LOCA. Therefore, NuScale is proposing that limitations be applied that restrict application of the ECCS evaluation model to a subdomain of components and phenomena that are addressed by the Appendix K rules. This is possible since only a subset of the phenomena that are addressed in Appendix K will be encountered in design-basis LOCAs due to the simple and safe design of the NuScale Power Module. Each requirement of Appendix K has been addressed with a discussion on how the requirement will be met. Phenomena not addressed by the Appendix K will be treated conservatively by the NuScale EM.

Appendix A. NRELAP5 Loss-of-Coolant Accident Test Plan

The table below lists the NRELAP5 validation matrix cases that are currently planned to be documented in the LOCA methodology report to establish the applicability of the NRELAP5-based EM for NuScale Power Module ECCS performance analysis.

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